ACCESSION #: 9902260093

NON-PUBLIC?: N

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Brunswick Steam Electric Plant (BSEP), PAGE: 1 OF 6

Unit No. 1

DOCKET NUMBER: 05000325

TITLE: Insertion Of Manual Reactor Trip Due To Reactor Vessel

Bottom Head Stratification

EVENT DATE: 01/23/1999 LER #: 1999-002-00 REPORT DATE: 02/22/1999

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 25

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: Steven F. Tabor, Project Analyst - TELEPHONE: (910) 457-2178

Regulatory Affairs

COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE TO EPIX:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On January 23, 1999, at 0638 hours, with Unit 1 reactor power at approximately 25 percent, a manual reactor trip was inserted. Prior to the insertion of the manual reactor trip, operation with a single loop of the reactor recirculation system resulted in reactor

vessel bottom head stratification and operation near the vessel pressure and temperature limits. Based on these conditions, the decision was made to insert the manual reactor trip signal. Reactor water level decreased below the reactor water low level 1 trip setpoint (i.e., 166 inches), resulting in the actuation of Primary Containment Isolation System (PCIS) Groups 2, 6, and 8 (i.e., isolation of the Drywell Floor and Equipment Drains, Containment Atmospheric Control system valves and the Shutdown Cooling system). Plant systems responded as designed. The causes of the event are attributed to (1) the failure to adequately evaluate the impact of a recently installed thermal hydraulic instability modification on plant operations and (2) the lack of awareness by control room personnel of the effects of longer than anticipated single loop operations at minimum flow conditions and the impact of such conditions on reactor vessel bottom head stratification. Corrective actions include establishing the necessary procedural controls and training to preclude recurrence. This event is being reported in accordance with the requirements of 10 CFR 50.73 (a)(2)(iv) in that the condition resulted in the manual and automatic actuation of an Engineered Safety Feature systems.

INITIAL CONDITIONS

On January 23, 1999, a downpower of the Unit 1 reactor was in progress to support planned maintenance activities. As part of the planned maintenance, replacement of worn brushes on both reactor recirculation (RR) system [AD] pump motor generator (MG)/[MG] sets was scheduled to be performed. The work sequence included decreasing reactor power to approximately 35 percent and securing the 1A and 1B RR MG sets one at a time to support MG set brush replacements.

EVENT NARRATIVE

On January 23, 1999, at 0201 hours, with reactor power reduced to approximately 38 percent, operators secured the RR system pump 1A. This action resulted in an unexpected entry into the thermal hydraulic instability (THI) restricted region on the power-to-flow map and a fraction of core boiling boundary (FCBB) value of greater than 1. The required action as delineated in the Limiting Condition for Operation in Technical Specification (TS) 3.2.3, Fraction of Core Boiling Boundary (FCBB), was initiated to restore FCBB to within limits within 2 hours. To exit the THI restricted region, operators increased core flow by increasing the RR system pump 1B speed in accordance with the applicable operating procedure. By 0210 hours, the restricted region was exited. RR system pump 1A was then placed under clearance and brush replacement performed on the MG set. During performance of the 1A RR MG set brush replacement, operators inserted control rods to prevent reentry into the TFU restricted region in anticipation of restarting the MG set. At 0408 hours, after completing the 1A RR system pump maintenance activity, operators, in accordance with procedure requirements for restart of the 1A RR system pump, decreased 1B RR system pump flow from 28,000 gallons per minute (gpm) to approximately 24,000 gpm. Minor feedwater flow oscillations in feedwater flow increased as flow was reduced to 24,000 gpm.

At approximately 0428 hours, the combination of the feedwater flow oscillations and reduced power operation resulted in an automatic runback of the 1B RR system pump when the total feedwater flow decreased below the 20 percent total feedwater flow setpoint. Following the runback, RR system pump 113 flow stabilized at approximately 15,000 gpm with reactor power at approximately 25 percent. The resulting values of core flow and reactor power were slightly below the THI restricted region on the power-to-flow map.

As a result of a modification which had been recently installed to address THI concerns, the average power range monitor (APRM) [IG/MON] rod block setpoint bad been lowered at the existing flow rate of 15,000 gpm from approximately 71 percent reactor power to 26 percent reactor power. At 25 percent reactor power, the 10 percent margin to the APRM rod block setpoint, that was required by procedure to support RR system pump 1A restart, was not available. Reactor power level and core flow were maintained while methods to restart the 1A RR system pump were investigated by Operations personnel.

With reduced core flow conditions, thermal stratification began to occur in the reactor vessel bottom head region. In addition, the reactor vessel pressure and temperature values were approaching the limits specified in TS 3.4.9, RCS Pressure and Temperature (P/T) Limits. Based on these conditions and the inability to meet the conditions for returning the RR system pump 1A to service, operations manually inserted a reactor trip at 0638 hours. Subsequent to the reactor trip, during investigation into the event, it was determined that the reactor vessel bottom head experienced a maximum cooldown rate of approximately 150 degrees Fahrenheit (F) per hour from 0430 to 0530 hours following the runback of the 1B RR system pump and a maximum heat-up rate of approximately 250 degrees F per hour from 0638 to 0738 hours following the reactor trip. In addition, it was determined that the RCS P/T limits were momentarily exceeded until insertion of the reactor trip restored compliance with applicable P/T limits.

Following the insertion of the manual reactor trip signal, reactor water level decreased to approximately 160 inches which is below the low level 1 trip setpoint of 166 inches. This level transient is normal and was anticipated prior to insertion of the manual reactor trip. Primary Containment Isolation system (PCIS)/[JM] isolation groups 2, 6, and 8 actuations (i.e., isolation of the Drywell Floor and Equipment Drains [JM], Containment Atmospheric Control system [IK] valves and the Shutdown Cooling system [BO]) were received, as expected, following the reactor trip. Plant systems responded as designed. By 0912 hours, the PCIS isolation signals were reset. At 0945 hours, a notification to the NRC (i.e., Event Notification #35310) was made in accordance with the requirements of 10 CFR 50.72(b)(2)(ii), in that the condition resulted in the manual and automatic actuation of an Engineered Safety Feature systems [JE].

While the unit was shutdown, the brushes on the 1B RR MG set were replaced. By 1235 hours, the 1B RR MG pump was returned to service. At 2245 hours, in accordance with TS 3.4.9 Action A.2 an evaluation to determine whether the RCS was acceptable for continued operation was completed and the results reviewed with Operations to support declaring the RCS operable. The evaluation concluded that the RCS was acceptable for continued operation. On January 24, 1999, at 0448 hours, following the verification of unit system readiness, reactor startup was commenced.

This event is being reported in accordance with the requirements of 10 CFR

50.73 (a)(2)(iv) in that a manual and automatic actuation of Engineered Safety Feature systems occurred.

EVENT CAUSE

Evaluation of this event identified two root causes, (1) inadequate evaluation of the THI modification impact on plant operation and (2) a lack of operator knowledge concerning the effects of single loop/minimum now operation on reactor vessel bottom head stratification. Discussion of each of the causes is provided below:

Cause 1:

Site personnel failed to adequately evaluate the impact of the THI modification on plant operations. The THI modification had been declared operable in July of 1998. The modification installed instrumentation which provides the operator with an indication of reactor core stability performance based on neutronic/thermal hydraulic parameters. The individuals responsible for development and review of THI modification related procedures did not identify the 10 percent margin to the APRM rod block setpoint requirement. In addition, although the operator training developed to support the THI modification implementation addressed modification function and impact to power to flow curves, rod block and scram setpoints, and transients, single loop operation or transient recovery was not addressed in the training. The transients evaluated on the simulator included the plant response to RR system and feedwater system pump trips and RR system pump runbacks. However, the THI modification acceptance testing did not include any simulated operator action required for recovery.

Cause 2:

Control room personnel were unaware of the effects of longer than anticipated single loop operations at minimum flow and its impact on reactor vessel bottom head stratification. Bottom head temperature stratification is a concern that had previously been identified on the loss of a RR system pump; however, the concern centered around the loss of flow from the bottom head area due to the bottom head drain being clogged. A clogged bottom head drain would mean that bottom head drain temperature, could not be obtained, resulting in the inability to determine the delta between reactor vessel top head and bottom head temperatures to allow for a RR system pump restart. The potential for stratification in the reactor vessel bottom head with one RR system pump running had not been identified as a concern and consequently, procedures and training had not been established to ensure proper monitoring and operation in this condition.

CORRECTIVE ACTIONS

As an interim measure, a Standing Instruction was established to address plant operation while in single loop until additional procedure guidance is implemented.

Revisions to appropriate Operations procedures have been implemented to remove the 10 percent APRM rod block/reactor power margin for RR pump restart and to provide guidance for (1) conducting a unit

shutdown while in RR system single loop operation and (2) monitoring bottom head temperature for indications of stratification during low RR system pump flow conditions.

The appropriate Nuclear Engineering procedures have been reviewed to verify proper guidance is provided for operation with TFU restrictions. No additional revisions to these procedures were needed.

A review will be performed of the complex integrated plant modifications which are scheduled for implementation in the upcoming Unit 2 B214R1. refuel outage to ensure proper training and procedure revisions have been identified.

Additional training on single loop operation and recovery, and the THI modification will be conducted with licensed Operators during Phase 2 of the 1999 licensed Operator Retraining program.

A review of the plant modification process will be performed to verify that:

. Complex modifications are evaluated on the simulator for operational impact prior to installation in the plan

. Planning and scheduling of modification implementation is addressed

. Initial and follow up training are evaluated

Any corrective actions identified during this review will be tracked in accordance with the corrective action program.

SAFETY ASSESSMENT

The safety significance of this occurrence is considered minimal as operation of the plant was within design limits and the affected systems responded as designed. In addition, the Emergency Core Cooling systems were operable during the event. As previously indicated, the evaluation of the effects of out-of-limit conditions on the reactor vessel structural integrity determined that the RCS was acceptable for continued operation.

PREVIOUS SIMILAR EVENTS

A review of LERs submitted within the last three years identified no previous similar occurrences.

COMMITMENTS

A review will be performed by March 17, 1999, of the complex integrated plant modifications which are scheduled for implementation in the upcoming Unit 2 B214R1 refuel outage to ensure proper training and procedure revisions have been identified.

Additional training on single loop operation and recovery, and the THI modification will be conducted by April 30, 1999, with Licensed Operators during Phase 2 of the 1999 Licensed Operator Retraining program.

A review of the plant modification process will be performed by June 17, 1999, to verify that:

. Complex modifications are evaluated on the simulator for operational impact prior to installation in the plant

. Planning and scheduling of modification implementation is addressed

. Initial and follow up training are evaluated

Energy Industry Identification System (EIIS) codes are identified in the

text as [xx].

ATTACHMENT 1 TO 9902260093 PAGE 1 OF 2 ATTACHMENT 1 TO 9902260093 PAGE 1 OF 2

CP&L

Carolina Power & Light Company

P.O. Box 10429

Southport, NC 28461-0429

FEB 22 1999

SERIAL NO: BSEP 99-0024 10 CFR 50.73

U. S. Nuclear Regulatory Commission

ATTN: Document Control Desk

Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1

DOCKET NO. 50-325/LICENSE NO. DPR-71

LICENSEE EVENT REPORT 1-1999-002-00

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73,

Carolina Power & Light Company submits the enclosed Licensee Event Report.

This report fulfills the requirement for a written report within thirty

(30) days of a reportable occurrence.

Please refer any questions regarding this submittal to Mr. Keith R. Jury, Manager - Regulatory Affairs, at (910) 457-2783.

Sincerely,

h Plant General Manager Brunswick Steam Electric Plant

Sincerely,

Jeffrey J. Lyash

Plant General Manager

Brunswick Steam Electric Plant

SFT

Enclosure: Licensee Event Report

Document Control Desk

BSEP 99-0024/Page 2

cc:

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